November 13, 2002

MEMORANDUM TO: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

FROM: Farouk Eltawila, Director ***/RA/*** original signed by C. Trottier
Division of Safety Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: SUMMARY OF THE PUBLIC WORKSHOP ON KEY ISSUES RELATED TO THE LICENSING OF FUTURE NON-LIGHT WATER REACTORS

On October 22 and 23, 2002, the Office of Nuclear Regulatory Research (RES) at the Nuclear Regulatory Commission (NRC) held a public workshop in Rockville, MD, to address the seven policy issues identified in SECY-02-0139 related to the licensing of future non-light water reactors (non-LWRs). A Federal Register notice (67 FR 60702) titled "Workshop on Key Issues Related to the Licensing of Future Non-Light Water Reactors" was posted to announce the workshop background and agenda.

In addition to the agenda, presentation slides, and instructions for written comments, the following documents were distributed at the workshop: SECY-02-0139 "Plan for resolving policy issues related to licensing non-light water reactor designs, SECY-93-092 "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and Candu 3 designs and Their Relationship to Current Regulatory Requirements," a staff requirements memo (SRM) on SECY-93-092 dated 7/30/93, ACRS letter of 2/19/93 addressing SECY-93-092, and the NRC staff response to the ACRS letter of 2/19/93 in a letter dated 3/24/93. These documents were made available as they are older and may be difficult to access using ADAMS.

The purpose of the workshop was to gain stakeholder feedback on each of the policy issues to aid in developing the staff recommendations due to the Commission in December, 2002. The NRC noted that it was not proposing a change to the Safety Goal Policy. Additionally, a risk-informed, performance-based approach was to be used wherever practical. Each issue was to be discussed from a technology-neutral standpoint while considering implications for non-light water reactors (LWRs). The NRC noted that the issues discussed relate to reactor design and operation and that security, safeguards, and fuel cycle issues were outside of the scope of the workshop.

This summary highlights the main issues discussed at the workshop. NRC presented the background, key questions, and options associated with each issue (see Attachment 5) prior to opening the floor to discussion on the issue. Many stakeholders noted that these issues are inter-related and should be addressed together to achieve the desired goals. The discussion held on each issue is detailed in the remainder of this summary.

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List of Attachments

1. Agenda
2. List of Attendees
3. Instructions for Written Comments
4. "Future Reactor Activities at the NRC"
5. "NRC Workshop on Key Issues Related to the Licensing of Future Non-Light Water Reactors," Presentation Slides
6. Letter from Laurence L. Parme, Manager, GT-MHR Safety and Licensing of General Atomics re: "Comments on SECY-02-0139 Related to Licensing Non-Light Water Reactor Designs"
7. Fleming, Karl N. and Silady, Fred A. "A Risk Informed Defense-in-Depth Framework for Existing and Advanced Reactors. "Draft - expected publishing date December 2002 in *The Journal of Reliability Engineering and System Safety."*
8. Letter from Dr. Per Peterson, Department of Nuclear Engineering at the University of California, Berkeley, to Dr. Farouk Eltawila, Director of Division of Systems Analysis and Regulatory Effectiveness in the Office of Research at the NRC re: "Workshop on Key Issues Related to the Licensing of Future Non-Light Water Reactors–Discussion on Issue 6."

Enhanced Safety

The question of how to implement the Commission's expectations for enhanced safety was posed to the workshop. The 1985 Severe Accident Policy Statement said that new plants are to achieve a higher standard of severe accident safety performance than prior designs. The 1986 Advanced Reactor Policy Statement said that as a minimum, the same degree of protection that is required for current generation LWRs is expected for new plants, but that the Commission expected enhanced margins of safety.

A number of stakeholders voiced that the NRC should not require enhanced safety, as the current regulations provide for an acceptable level public health and safety. It was stated numerous times that the Commission should expect enhanced safety, however they should not regulate it. There was uncertainty regarding what enhanced safety means, and it was argued that different plant designs should have the flexibility to demonstrate enhanced safety in different ways.

While the stakeholders voiced that the expectations stated in the Advanced Reactor Policy Statement would be met without regulations mandating enhanced safety, the NRC stated that it had a history of having difficulty enforcing policy statements. Additionally, the NRC questioned whether it would be detrimental to public confidence if the regulator relies on the industry to provide enhanced safety. Stakeholders voiced that the licensing of non-LWR advanced reactors should follow the advanced light water reactor (ALWR) certification model, where an applicant proposed enhanced safety features which NRC staff (with Commission endorsement) reviewed and supplemented.

There was discussion on developing the right metrics and criteria for what enhanced safety means for non-LWRs. No conclusion were reached on what was meant by an enhanced level of confidence and fewer uncertainties. Many stakeholders present agreed that it is reasonable

to require additional oversight or testing on key areas where experience is limited and/or uncertainties are high.

There was discussion on whether the Commission's Safety Goals should be applied on a per plant or a per site basis. The key issue in considering a risk-informed approach, given the safety goals, hinges on the effect of multiple plants on a site. No general consensus was reached on this point.

Defense in Depth

The question of whether or not to clearly define the elements of defense in depth (DID) was posed to the workshop. Additionally, the workshop was asked to provide their input of what the elements should be, i.e. programmatic, process, physical, or a combination. DID is a concept that has been referenced in Commission policies. However, the NRC has not articulated what the elements of DID are beyond a brief definition in their 1999 white paper on risk-informed, performance-based regulation, and a concept description in 10 CFR 50, Appendix R (Fire Protection). Definitions of DID have been presented in works such as the Nuclear Energy Institute (NEI) 02-02 white paper on Risk-Informed, Performance Based Regulatory Framework for Power Reactors, the PBMR paper on DID in *Reliability Engineering and System Safety* (see Appendix), and the International Atomic Energy Agency (IAEA) publication INSAG-10.

There was general agreement that DID needs to be defined by the NRC. Industry representatives voiced a preference for programmatic and process-oriented, or functional, elements of DID, their reasoning being that physical barriers would arise from this approach. Workshop participants also discussed the relationship between DID and probabilistic risk assessment (PRA). Some participants voiced that DID should not contradict PRA results. Others suggested that DID should be used to compensate for weaknesses in PRA analysis.

The NRC stated that if there is a decision to define DID, the staff would interact with the stakeholders in preparing the definition so as to gain further input.

International Codes and Standards

The question of how NRC requirements for non-LWRs should relate to international safety standards and requirements was posed to the workshop. This question has become pertinent in recent years as design and marketing efforts have become more international in nature. Additionally, NRC's experience in and infrastructure for licensing non-LWRs is limited. NRC Management Directive 6.5, "NRC Participation in the Development and Use of Consensus Standards," encourages the NRC to utilize consensus standards-where practical.

The workshop participants agreed that the NRC must be involved at some level with international standards because of the international nature of new reactor designs. It was pointed out that replacement parts designed to non-US codes and standards are being ordered from other countries for currently operating US plants. This brings into issue the need of the NRC to be cognizant of manufacturing standards, fire codes, and other state laws that affect the use of codes and standards.

There was general agreement that the NRC should review any international codes and standards referenced in applications and pre-applications. Additionally, stakeholders agreed that the NRC should utilize the experience of other regulatory bodies to facilitate the review of

international codes and standards. This recommendation came with the cautionary note that the NRC should not 'rubberstamp' a standard simply because another regulatory body endorsed
it. The workshop participants agreed that the NRC should be proactive in selected areas by participating in development of and endorsing international codes and standards that could fill critical gaps in the infrastructure for non-LWRs.

Event Selection
The question of to what extent a probabilistic approach can be used to establish the licensing basis in terms of event selection was posed to the workshop. Previous Commission guidance was provided in the form of a Staff Requirements Memorandum (SRM) dated July 30, 1993. The Commission endorsed the use of deterministic analysis supplemented with PRA insights to select events, and to categorize the events according to their expected frequencies, as well as to deterministically select challenging events for the assessment of source term, containment evaluation, and safety margins.

The workshop participants agreed that event categories should be defined probabilistically. A preference for using conservative analysis to analyze design basis events (DBEs)and a best-estimate analysis to analyze beyond DBEs was voiced. Additionally, the participants endorsed replacing the single failure criterion with reliability criterion. Some workshop participants stated a need for a document articulating how to apply PRA to non-LWRs.

Source Term
The question of under what conditions a scenario-specific accident source term should be used for licensing decisions was posed to the workshop. Previous Commission guidance in the form of a July 30, 1993 SRM approved the use of scenario specific source terms provided there was sufficient understanding of plant and fuel performance. This guidance also instructed that events selected for source term evaluation should bound design dependent uncertainties and severe accidents.

The workshop participants felt that although a bounding source term is used in the licensing of current LWRs, the regulations do allow for either approach. Workshop participants felt that the regulations should retain the flexibility of allowing the use of either a bounding or a scenario-specific source term. Additionally, some voiced that there was no need to modify the Commission decisions of 1993.

Many workshop participants stated that a core melt accident is not the correct metric for establishing a severe accident source term for gas-cooled reactors. A comment was made that accidents at the THTR and the AVR in Germany should be investigated by the NRC prior to deciding that a core melt accident should not be used.

Some participants felt that the NRC should publish a document for non-LWRs that mirrors NUREG-1465 "Accident Source Terms for Light-Water Nuclear Power Plants". Other participants felt that the NRC should develop a NUREG to determine the bounding source term for each new reactor design.

Containment vs. Confinement

The question of when a plant can be licensed without a pressure retaining containment was posed to the workshop. Previous Commission guidance in the form of a July 30, 1993 SRM stated that the building must be robust enough such that radionuclide release limits specified in 10 CFR 100 are met. Additionally, the guidance mandated that for 24 hours following the onset of core damage, the limiting building leak rate assumed in the accident evaluation shall not be exceeded, and that after 24 hours no uncontrolled release of radioactivity could occur. Many workshop participants agreed that the requirements for reactor containment or confinement should be defined functionally, so that the uncontrolled release of radionuclides to the environment is prevented. The reasoning for this was that the NRC should not be concerned with the type of building, rather with the effect of the building. The NEI 02-02 white paper titled "Risk-Informed, Performance Based Regulatory Framework for Power Reactors" was cited by some participants as providing functional requirements that the NRC may want to consider.

No general consensus on this issue was attained at the workshop. Proponents of confinement said that for some non-LWR designs, a pressure-retaining containment actually reduces safety. Some participants said the NRC should be regulating such that the public is adequately protected and that any design meeting release limits should be allowed. They also said that defense in depth does not specify a certain type of building.

Proponents of containment were concerned about the accuracy of PRA and violation of defense in depth principles. Some said that unforseen DBAs may occur, especially with designs that have limited operational experience, and that these unforseen accidents mandate the need for a containment. Additionally, they stated that containments are integral for upholding public confidence in reactors.

Emergency Preparedness

The question of under what conditions the emergency planning zone (EPZ) can be reduced, including a reduction to the exclusion area boundary (EAB) was posed to the workshop. Previous Commission guidance in the July 30, 1993 SRM stated that it was premature to revise emergency planning for advanced reactors, but that NRC staff should remain open to suggestions to simplify EP for reactors that are designed with greater safety margins.

There was general consensus that in the near term this issue is a moot point as all early site permits (ESPs) are for existing sites. While comments were made saying that it would be useful to identify the history of the current 10 mile EPZ radius, it was also noted that current regulations are able to accommodate HTGRs (e.g. Fort St. Vrain was licensed with a 5 mile EPZ) and that this issue should be deferred for now.

Attachments: As stated

cc w/ atts. (via e-mail)
PMNS
JWermiel
SNewberry
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EWilliamson, OGC

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Attachments: As stated
cc w/ atts. (via e-mail)
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REAHFB R/F
DSARE R/F

AThadani/JStrosnider,	RES GHolahan, NRR	JLyons, NRR
FEltawila, RES	RBarrett, NRR	
MMayfield, RES	DMatthews, NRR	
Advanced Reactor Group, RES (25)	CGrimes, NRR	
JJohnson, RES	MJohnson, NRR	
CAder, RES	LFields, NRR	
PNorian, RES	GBagchi, NRR	
JMoore, OGC	SCollins, NRR	**Package ML023180767**

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DATE	11/12/02*	11/13/02*	11/13/02*

Workshop on Policy Issues Associated with Non-LWR licensing
(Double Tree Hotel - Regency Room)

AGENDA

Tuesday (October 22, 2002)

1:00 - 1:15 - Introduction

1:15 - 1:45 - Overview of Issues and Plans for Resolution

1:45 - 3:45 - Expectations for Safety

3:45 - 4:00 - BREAK

4:00 - 5:00 - Defense-in-Depth

Wednesday (October 23, 2002)

9:00 - 10:00 - Use of International Codes and Standards

10:00 - 10:15 - BREAK

10:15 - 11:15 - Event Selection and Safety Classification

11:15 - 12:30 - LUNCH

12:30 - 1:30 - Licensing Source Term

1:30 - 2:30 - Containment vs. Confinement

2:30 - 2:45 - BREAK

2:45 - 3:45 - Emergency Preparedness

3:45 - 5:00 - General Discussion, Summary and Wrap-up

Attachment 1

Workshop to Discuss Key Issues Related to the Licensing of Future Non-Light Water Reactors
October 22-23, 2002
Attendance List

Last Name	*First Name*	*Affiliation*	*Phone Number*	*Email Address*
Bagchi	Goutam	NRC/NRR	301-415-3298	gxb1@nrc.gov
Bell	Russ	Nuclear Energy Institute	202-739-8087	rjb@nei.org
Braun	Joseph C.	Argonne National Lab	630-252-5574	joebraun@anl.gov
Browde	Shana	NRC/RES	301-415-7652	srb1@nrc.gov
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Eltawila	Farouk	NRC/RES	301-415-7499	fxe@nrc.gov
Feltus	Madeline A.	DOE/NE	301-903-2308	madeline.feltus@hq.doe.gov
Grimes	Chris	NRC/NRR	301-415-1282	cig@nrc.gov
Hay	Lane	SERCH Bechtel	301-228-6312	hlhay@bechtel.com
Heymer	Adrian	Nuclear Energy Institute	202-739-8094	aph@nei.org
Holbrook	Mark	INEEL	208-526-4362	hbk@inel.gov
Kida	Muneo	Japan NOS		kidax@janus.co.jp
King	Tom	NRC/RES	301-415-6345	tlk@nrc.gov
Langman	Vince	AECL Technologies Inc.	905-823-9060x6543	langmanv@aecl.ca
Mays	Gary	Oak Ridge National Lab	865-574-0394	maysgt@ornl.gov
McKenna	Eileen	NRC/NRR	301-415-2189	emm@nrc.gov
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Quint	Matthew	Embassy of Australia	202-797-3043	matthew.quint@austemb.org
Reid	Cal	Bechtel	301-228-6533	creid@bechtel.com
Riccio	Jim	Greenpeace	202-319-2487	jim.riccio@wdc.greenpeace.org

Ridgely	John N.	NRC/RES	301-415-6555	jnr@nrc.gov
Ritterbusch	Stan	Westinghouse Electric	860-731-6621	stanley.e.ritterbusch@us.westinghouse.com
Settles	Alwyn C.	Illinois Department of Nuclear Safety	217-557-4618	settles@idns.state.il.us
Shahrokhi	Farshid	Framatome-ANP	434-832-2923	f.shahrokhi@framatome.ANP
Tripathi	Raji	NRC/RES	301-415-7472	rrt1@nrc.gov
Weil	Jenny	McGraw-Hill	202-383-2161	jenny_weil@platts.com
Williams	Mark	DOE/EH	301-903-6892	mark.williams@hq.doe.gov
Wilson	Jerry	NRC/NRLPO	301-415-3145	jnw@nrc.gov
Zinke	Geore	Entergy	601-368-5381	gzinke@entergy.com

Written Comments

If written comments are to be provided, please send then by November 5, 2002 to:

- e-mail - FXE @NRC. GOV
- U.S. mail - Dr. Farouk Eltawila, Director
  Division of Systems Analysis and
  Regulatory Effectiveness
  Office of Nuclear Regulatory Research - Mail
  Stop T-10E32
  U.S. Nuclear Regulatory Commission
  Washington D.C. 20555-0001

Future Reactor Activities at NRC

LWRs:

- **AP-1000 - design certification review underway**
- **ESBWR - pre-application review anticipated**
- **SWR-1000 - pre-application " "**
- **ACR-700 - pre-application " "**
- **IRIS - pre-application " "**

- **Non-LWRs:**
  - **GT-MHR - pre-application review underway**
  - **DOE-Generation IV - NRC observer**
  - **PBMR - pre-application review suspended**

- **Early Site Permits**
  - **3 applications expected in 2003:**
    - **existing sites**
    - **multiple units**

Attachment 4

NRC Workshop on Key Issues Related to the Licensing of Future Non-Light Water Reactors

October 22-23, 2002

Attachment 5

Introduction

- SECY-02-0139, “Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactors Designs” identified seven issues with potential policy implications resulting from the preapplication reviews to date of the Pebble Bed Modular Reactor (PBMR)and the Gas Turbine-Modular Helium Reactor (GT-MHR):

  - Expectations for safety
  - Defense-in-depth
  - Use of international codes and standards
  - Event selection
  - Source term
  - Containment vs. Confinement
    Emergency preparedness

Introduction (Cont.)

- Recommendations on their resolution are due to the Commission in late December 2002.

- Purpose of this workshop is to:
  - discuss the origin and background for each issue
  - discuss key questions associated with the issues
  - discuss options for their resolution
  - discuss advantages and disadvantages of the options

- Scope of issues is reactor design and operation:
  - security, safeguards and fuel cycle issues are not the scope of this workshop

Introduction (Cont.)

- Many of the issues are linked.
- Four of the issues were presented to the Commission 10 years ago (SECY-93-092) and the Commission provided guidance in a staff Requirements Memorandum (SRM) of July 30, 1993.
  - appropriate to revisit these issues given the current emphasis on risk-informed regulation and the Commission's Strategic Plan.
- Three new overarching issues resulting from various factors associated with the designs.
- Potential implications for future LWRs.

Evaluation of Issues

In assessing the options and developing the recommendations on the seven issues the following general guidelines are proposed:

- The risk to the population around a nuclear power plant site should be consistent with the Commission's Safety Goal Policy.

- The Commission's Strategic Plan (NUREG-1614) performance goals of maintaining safety; enhancing public confidence; efficiency, effectiveness and realism; and reducing unnecessary burden should be used to assess the advantages and disadvantages of the options and develop recommendations.

- A risk-informed and performance based approach should be chosen, wherever practical.

Evaluation of Issues (Cont.)

- A technology neutral approach should be used.
- The implications for LWRs need to be considered.
- The practicality of the recommendations (i.e., resources, schedule and benefit) need to be considered.

Expectations for Enhanced Safety

- **Issue: How to implement the Commission's expectations for enhanced safety:**

  – 1985 Severe Accident Policy Statement (new plants are to achieve a higher standard of severe accident safety performance than prior designs).

  – 1986 Advanced Reactor Policy Statement (as a minimum, the same degree of protection that is required for current generation LWRs. Expectation of enhanced margins of safety).

  – June 15, 1990 SRM on Safety Goal Implementation - (don't use industry's objectives to establish new requirements).

  – ALWR Design Certifications (case-by-case enhancements).

  – NRC Strategic Plan – NUREG-1614 (maintain safety).

Enhanced Safety (Cont.)

- **Key Questions:**
  - How should the potential for additional plants (nationwide and on a per site basis) influence the level of safety required for future plants?
  - Should the Commission's Safety Goal Policy apply on a per plant or per site basis?
  - How should the Commission's performance goal to "maintain safety" impact the level of safety required for future plants?
  - Should enhanced accident prevention be required to help compensate for the larger uncertainties of non-LWRs, particularly these associated with severe core damage accidents?

Enhanced Safety (Cont.)

- **Key Questions (Cont.):**
  - What enhancements should be considered to compensate for uncertainties
    - safety margins?
    - testing?
    - oversight?
  - Implications for future LWRs?

- **Options:**
  - Require current level of safety
    - with expectation that applicants will provide enhanced safety
  - Require enhanced level of safety
    - e.g., more stringent CDF

Enhanced Safety (Cont.)

- **Options (Cont.):**
  - Require enhanced level of confidence
    - e.g., additional testing/additional oversight
  - Encourage industry to implement enhanced safety
- **Advantages of Requiring Enhanced Safety:**
  - helps compensate for less experience
  - helps compensate for integrated risk of multiple units
  - public confidence
- **Disadvantages of Requiring Enhanced Safety:**
  - could imply current plants are not safe enough
  - could result in dual set of regulatory requirements

Defense-in-Depth (DID)

- **Issue: How to specify DID for non-LWRs**
  - Mentioned in Commission policies, but no articulation as to the elements of DID
  - Concept described briefly in 10CFR 50, Appendix R (Fire Protection)
  - Commission definition of DID provided in 1999 risk-informed, performance-based regulation white paper
  - IAEA and INSAG have published descriptions of DID
- **Key Questions:**
  - Would a more comprehensive description of DID be useful?
    - for reactor designers?
    - for other activities?

Defense-in-Depth (DID) (Cont.)

- **Key Questions (Cont.)**:
  - What are the elements of DID:
    - programmatic (e.g., QA, EP, ISI, etc.)
    - physical (e.g., barriers, redundancy, etc.)
    - process (e.g., treatment of uncertainties, safety not reliant on single element)
    - prevention vs. mitigation
    - structuralist vs. rationalist approach
  - Would the revised reactor oversight program cornerstones be a good DID structure?
- **Options**:
  - Case-by-case determination, depending upon:
    - Plant Design
    - Uncertainties

Defense-in-Depth (DID) (Cont.)

- **Options (Cont.)**:
  - Develop description or policy statement articulating specific programmatic and/or physical elements of DID
  - Develop description or policy statement articulating DID as a process

- **Advantages of developing a description of DID:**
  - Provides guidance on how to implement the DID philosophy discussed in Commission policies.
  - Could form the foundation for a future plant licensing framework
  - Could be useful in areas other than reactor licensing (e.g., Regulatory Analysis)

- **Advantages of applying a case-by-case determination of DID:**
  - details of DID are too design dependent

International Codes and Standards

- **Issue: How should NRC requirements for non-LWRs relate to international safety standards and requirements?**
  - international nature of design and marketing efforts
  - less NRC experience and infrastructure for non-LWRs
  - design and safety consensus standards

- **Key Questions:**
  - Applicability of NRC Management Directive 6.5 “NRC Participation in the Development and Use of Consensus Standards”?
  - How should NRC decide on the extent to which it should participate in international codes and standards activities?
  - What international codes and standards activities should NRC participate in:
    - IAEA safety standards?
    - ISO?
    - Others?

International (Cont)

Key Questions (Cont.)

- To what extent should NRC meet international safety standards?
- To what extent should NRC attempt to harmonize with other nuclear safety regulatory organizations?

- **Options:**

  - No specific relation
    - Review international codes and standards on an as necessary basis as part of an applicant's licensing submittal
  - Review and endorse existing international codes and standards, whenever practical
  - Participate in the development of international codes and standards and endorse, whenever practical
  - Attempt to harmonize requirements with other regulatory bodies

International (Cont)

- **Advantages of utilizing international codes and standards:**
  - Can help fill gaps in NRC infrastructure
  - NRC benefits from international experience
  - Consistent with Management Directive 6.5
  - Public confidence?
- **Advantages of not utilizing international codes and standards:**
  - can focus resources on what applicants present for review
  - standards may be too general to be of much use?

Event Selection

- **Issue: To what extent can a probabilistic approach be used to establish the licensing basis:**
  - Event selection?
  - Safety classification?
  - Replace single failure criterion?

- **Previous Commission Guidance** (July 30, 1993 SRM):
  - Select events to be considered deterministically, supplemented with PRA insights.
  - Catagorize events according to expected frequency
  - Deterministically select challenging events for assessment of source term, containment evaluation and margins.

Event Selection (Cont)

- **Key Questions:**
  - Does a probabilistic approach go beyond the intent of the Commission's 1995 PRA Policy Statement?
  - What are the implications of bringing PRA more directly into the licensing basis:
    - PRA quality / completeness?
    - Document control?
    - Level of confidence?
  - How should PRA success criteria interface with DBA acceptance criteria?
  - What should be the criteria for:
    - event selection?
    - safety classification?

Event Selection (Cont)

- **Options:**
  - Use a deterministic approach supplemented by PRA
  - Use a probabilistic approach
  - Use a probabilistic approach, supplemented by engineering judgement
- **Advantages of probabilistic approach:**
  - provides more realism (i.e., integrates design, operation, human) in safety review
  - consistent with Commission's emphasis on risk-informed regulation
- **Disadvantages of probabilistic approach:**
  - larger uncertainties in non-LWR PRAs may make PRA less useful
  - could require more extensive R+D to provide data and models to support useable PRA

Source Term

- **Issue: Under what conditions should scenario specific accident source terms be used for licensing decisions?**

- **Previous Commission Guidance (July 30, 1993 SRM):**
  - approved the use of scenario specific source terms provided there is sufficient understanding of plant and fuel performance under normal and off-normal conditions
  - events selected for source term evaluation should bound design dependent uncertainties and severe accidents.

- **Key Questions:**
  - should a source term representative of severe core damage be a fundamental element of DID?
  - what level of confidence should be required in selecting source terms?

Source Term (Cont)

- **Options:**
  - Develop a deterministic bounding ST
  - Allow the use of scenario specific ST

- **Advantages of Using Scenario Specific Source Terms:**
  - adds more realism to analysis
  - consistent with PRA approach

- **Disadvantages of Using Scenario Specific Source Terms:**
  - puts large burden on understanding plant, fuel and fission product behavior over the life of the plant.

Containment vs. Confinement

- **Issue: Under what conditions can a plant be licensed without a pressure retaining containment building?**

- **Previous Commission Guidance (July 30, 1993 SRM)**
  - building must be adequate to meet radionuclide release limits
  - for 24 hours following the onset of core damage, the limiting building leak rate assumed in the accident evaluation shall not be exceeded. After 24 hours - no uncontrolled release of radioactivity.

- **Key Questions:**
  - should a pressure retaining building be a fundamental element of DID?
  - can the presence of a pressure retaining building have an adverse impact on safety?
  - what criteria should be applied to allow the use of a non-pressure retaining building?
  - what confidence level should apply?

Containment vs. Confinement (Cont)

- **Options:**
  - Require a pressure retaining building
  - Allow a design without a pressure retaining building
- **Advantages of a Pressure Retaining Building:**
  - less concern about fuel quality and performance
  - less concern about air or water ingress accidents in HTGRs or LMRs.
  - public confidence?
- **Disadvantages of a Pressure Retaining Building:**
  - provides less incentive for designers to stress accident prevention
  - allows pressure in building to act as mechanism to release radioactive material
  - could make passive decay heat removal systems more complicated

Emergency Preparedness

- **Issue: Under what conditions can the EPZ be reduced, including a reduction to the exclusion area boundary?**

- **Previous Commission Guidance (July 30, 1993 SRM):**
  - Premature to revise emergency planning for advanced reactors
  - Staff should remain open to suggestions to simplify EP for reactors that are designed with greater safety margins

- **Key Issues:**
  - Should some EP requirements be retained as a fundamental element of DID? If so, what should they be?
  - Should an assumed severe core damage accident be the basis for EP for non-LWRs?
  - What criteria should be used to allow a reduction in current EP requirements?
  - Should long plant response times be a factor in determining EP requirements?
  - Confidence Level?

EP (Cont)

- **Options:**
  - No reduction from current requirements
  - Allow a reduction in the EPZ
  - Allow a graded approach within the current EPZ

- **Advantages of reducing the EPZ:**
  - Provides incentive for designers to stress accident prevention

- **Disadvantages of reducing the EPZ:**
  - Public confidence?

GENERAL ATOMICS

U.S. Nuclear Regulatory Commission
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Washington D.C. 20555-0001
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5 November 2002

Subject: Comments on SECY-02-0139, Policy Issues Related to Licensing Non-Light Water Reactor Designs

Dear Dr. Eltawila:

On 22 and 23 November 2002, NRC held a workshop on Policy Issues Related to Licensing Non-Light Water Reactor Designs. At that time, members of the public including General Atomics provided comments on these issues. Several of these issues were explored in further detail in free and open discussions between the participants. At the end of the workshop, NRC summarized their understanding of the comments received.

General Atomics supports this effort to clarify and establish policy on these issues. We feel that they will be very important in future licensing of not only non-LWRs but very likely any next generation advanced reactors. In addition, we support the general direction the staff appears to be going in developing policies on these issues.

Because of the importance of these issues to us, we have summarized our comments made over the course of the 2-day workshop and are providing them to you in the attachment. Should you have any questions on this letter, the attachment, or this subject, please feel free to contact me at 858 443-2518.

Sincerely,

Laurence L. Parme
Manager, GT-MHR Safety and Licensing

Enclosures: (1)

Shana Browde,	Office of Nuclear Regulatory Research
Leslie Fields,	Office of Nuclear Reactor Regulation
Stewart Rubin,	Office of Nuclear Regulatory Research
Eugene Trager,	Office of Nuclear Regulatory Research
Madeline Feltus,	Department of Energy
Tom Miller,	Department of Energy

Attachment

Comments on Issues

7. Enhanced Safety

The expectation of enhanced safety stated in the commission's *Advanced Reactor Policy Statement* was not intended to lead to revised regulation mandating these enhancements. To the contrary, during the recent certification of three advanced light-water reactor designs, the commission made clear this was not its intent.

Nevertheless, we believe that enhanced safety as realized through greater design margins, reduced uncertainty, and/or the utilization of simplified, inherent, passive, or other means for safety function accomplishment is a reasonable expectation for advanced reactors. This expectation can be dealt with in a manner similar to that employed during the advanced light-water reactor certifications. Industry should be encouraged to provide for an increased level of safety through the application of risk-insights and severe accident mitigation features that are applicable to the specific design.

We have and continue to support performance-based metrics for judging safety. However, we caution against the blind application to non-LWRs of some of the existing metrics commonly used, specifically core damage frequency. In general, we find that the more closely a metric can be related to measurement of public health and safety the more useful and generally applicable it is.

Finally, General Atomics has believed that while the safety goals were intended for power reactor industry as a whole, the expectation is that new reactor facilities will meet these goals. Recognizing that a multiple reactor concept raises questions on how to quantify the risk, within the MHTGR preapplication submittals risk to the public was quantified for the site rather than on a per reactor basis. We continue to believe this is the proper way to assess risk for new plants – especially those utilizing multi-module design approaches such as the GT-MHR. Note that the assessment of risk on a site basis does not any way imply endorsement of an allocation of the risk goal up between reactors on a particular site.

8. Defense-in-Depth (DID)

General Atomics believes that all parties would be better served by an agreed upon definition of what is meant by defense-in-depth. Reference to several sources that might assist in developing such a definition was made during the workshop. In developing this definition we believe that it is important to

- Include the several aspects of defense-in-depth in the definition including design considerations, processes, and risk assessment based insights, and
- Keep the definition independent of design and instead focussed on what is being sought and what are the processes or programs to evaluate for defense-in-depth

Finally, we believe active stakeholder participation in developing this definition is required to ensure the success of the effort.

3. International Standards

The nuclear industry is becoming increasingly international with many suppliers now located solely offshore. General Atomics supports

- NRC review of international codes and standards submitted in an application,
- NRC's use of the experience gained by other regulatory bodies to facilitate the review of these international codes and standards,
- NRC's proactive involvement in specific and selected areas by participating and endorsing international codes and standards that are nonexistant in the US and required for non-LWRs – as determined by pre-application discussions, and

- NRC's consideration of reciprocity in the use of US codes and standards by foreign regulators.

5. Event Selection

General Atomics continues to support the sort of logical and systematic event selection methodology first proposed during the 1980s MHTGR preapplication review and more recently by Exelon Generation for the PBMR. We believe this or similar approaches have the benefits of;

- Utilizing the existing NRC review and licensing framework largely as is,
- Making use of modern mechanistic and probabilistic analysis techniques to comprehensively review and identify design specific phenomena and scenarios important to safety, and
- Providing greater predictability to the process.

4. Mechanistic Source Term

General Atomics believes that existing regulation already provides the flexibility to allow use of scenario specific, mechanistic source terms where sufficient data and certainty in accident phenomenology exists. Furthermore, the NRC position, stated in the 1993 SRM, is consistent with this. We see no reason why this position should be reconsidered. To the contrary, we believe the use of scenario specific, mechanistic source terms has enabled designers of advanced reactors to pursue different and highly promising approaches to safety that can only benefit public health and safety.

5. Containment v. Confinement

General Atomics has on repeatedly stated our view that this question cannot be addressed in isolation. Rather, we believe that the question can only be looked at and decided in the broader context of whether a design provides adequate assurance against uncontrolled release of radionuclides.

We believe that the overall function of radionuclide control is what should be assessed. On this basis the acceptability of particular retention features offered by various building designs can then be judged. We also acknowledge that the resolution of the question of what constitutes sufficient defense-in-depth may have a bearing on this issue.

6. Emergency Planning

General Atomics supports a "graded approach" to emergency planning, appropriate to the risk to the public. We believe that an approach in which the EPZ radius is determined based on bounding accidents scenarios is consistent with the background for the 10 mile EPZ used on existing reactors. DOE proposed such an approach for the MHTGR and we anticipate pursuing the same methods for establishing an EPZ for the GT-MHR. In the proposal emergency preparedness is retained and we are confident that the proposed approach provides a level of public protection at least as good as that offered by existing reactors.

We believe that current regulations are sufficiently flexible in the determination of the Emergency Planning Zone (EPZ) for non-LWRs. Therefore, we do not see that any changes in current regulations are required. However, we still require comment from NRC regarding the acceptability of the proposed method to determining EPZs.